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Reference: University of Maryland, Maryland University Training Reactor ("MUTR"), Docket No. 50-166, License No. R-70, Technical Specifications, Response to February 18, 2011, Request for Additional Information ("RAI") Regarding Remaining Technical Specifications

The University of Maryland herewith submits the following documents in connection with its application for a renewal of the MUTR license identified above:

1. Technical Specifications for the Maryland University Training Reactor, License Number R-70, Docket Number 50-166; Draft Edit in Response to USNRC Questions in Regard to MUTR Relicensing (April 26, 2011); and
2. Revisions to RAI Responses Submitted 31 January 2011 (3 pages); and
3. Responses to RAI Questions of 2-18-11 Meeting (2 pages); and
4. Response to the NRC Request for Further Information regarding the Dose to Workers, Member of the Public and the Nearest Residence during Normal Operations and during an MHA Incident at the MUTR, University of Maryland Training Reactor (5 pages).

There are two open issues with regard to the dose calculations performed by the Radiation Safety Office. The first concerns the maximum dose to the closest person in the event of an MHA with the fans off. We will revise the current Emergency Plan to require evacuation of the areas surrounding the reactor building, which will lower the maximum possible dose below 10CFR20 limits. We will work with appropriate University offices (e.g. Department of Environmental Safety, Department of Public Safety) to implement the EP revision.

The second issue is the maximum dose to the closest person during normal operation (fans off). As indicated in the dose calculation report, in order to have a reasonable value for the Ar-41 concentration in the building, we will measure the Ar-41 diffusing from the pool to the air by taking grab samples and counting them on a gamma spectroscopy system. These measurements will be done as soon as possible once the reactor is again operational – we are currently replacing the fission chamber so cannot run the reactor. A new fission chamber has been procured but has not yet been delivered.

If there are questions about the information submitted, please write me at 2309F Chemical & Nuclear Engineering Building, University of Maryland, College Park, MD 20742-2115 or email me at mohamad@umd.edu. I would appreciate it if you would copy Prof. Robert Briber on any correspondence: 2135 Chemical & Nuclear Engineering Building, University of Maryland, College Park, MD 20742-2115; rbriber@umd.edu.

A020
NRR

I declare under penalty of perjury that the foregoing and the enclosed documents are true and correct.

Sincerely,

Alsheikhly

Mohamad Al-Sheikhly
Professor and Director
Maryland University Training Reactor

Enclosures (4)

cc: Robert Briber

TECHNICAL SPECIFICATIONS

FOR THE

MARYLAND UNIVERSITY TRAINING REACTOR

License Number R-70

Docket Number 50-166

Draft Edit in Response to USNRC Questions in Regard to
MUTR Relicensing

26 April 2011

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Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.0 DEFINITIONS

1.1 ALARA (acronym for ``as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

1.2 CHANNEL - A channel is the combination of sensors, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a parameter.

1. Channel Calibration - A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

2. Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

3. Channel Test - A channel test is the introduction of a signal into the channel to verify that it is operable.

1.3 CONFINEMENT - Confinement means a closure on the overall facility that controls the movement of air into it and out, thereby limiting release of effluents, through a controlled path.

1.4 CORE CONFIGURATION – The core consists of 24 fuel bundles, with a total of 93 fuel elements, arranged in a rectangular array with one bundle displaced for the pneumatic experimental system; three control rods; and two graphite reflectors.

1.5 EXCESS REACTIVITY - Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff}=1)

1.6 EXPERIMENT - Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beamport or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be part of their design.

1. Routine Experiments - Routine Experiments are those which have been previously performed in the course of the reactor program.

2. Modified Routine Experiments - Modified routine experiments are those which have not been performed previously but are similar to routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiments.

3. Special experiments - Special experiments are those which are not routine or modified routine experiments.

1.7 EXPERIMENTAL FACILITIES - Experimental facilities are facilities used to perform experiments and include, for example, the beam ports, pneumatic transfer systems and any in-core facilities.

1.8 EXPERIMENT SAFETY SYSTEMS - Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.9 FUEL BUNDLE - The 4-rod bundle consists of an aluminum bottom, 4 stainless steel clad fuel elements and aluminum top handle.

1.10 FUEL ELEMENT - A fuel element is a single TRIGA fuel rod.

1.11 FULL POWER - Full licensed power is defined as 250 kW.

1.12 INSTRUMENTED ELEMENT - An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel.

1.13 ISOLATION – Isolation is the establishment of confinement, closing of the doors leading from the reactor bay area leading into the balcony area on the top floor, the door to the reception area on the ground floor, and the building exterior doors.

1.14 LIMITING CONDITIONS FOR OPERATION - Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

1.15 LIMITING SAFETY SYSTEM SETTING - Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

1.16 MEASURING CHANNEL - A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device, which are connected for the purpose of measuring the value of a variable.

1.17 MEASURED VALUE - The measured value is the value of a parameter as it appears on the output of a channel.

1.18 MOVEABLE EXPERIMENT - A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.19 ON CALL - A senior operator is available "on call" if the senior operator is either on the College Park campus or within 10 miles from the facility and can reach the facility within one half hour following a request.

1.20 OPERABLE - Operable means a component or system is capable of performing its intended function.

1.21 OPERATING - Operating means a component or system is performing its intended function.

1.22 REACTIVITY WORTH OF AN EXPERIMENT - The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

1.23 REACTOR CONSOLE SECURED - The reactor console is secured whenever all scrammable rods have been fully inserted and verified down and the console key has been removed from the console.

1.24 REACTOR OPERATING - The reactor is operating whenever it is not secured or shutdown.

1.25 REACTOR OPERATOR - A reactor operator (RO) is an individual who is licensed by the U.S. Nuclear Regulatory Commission (NRC) to manipulate the controls of the reactor.

1.26 REACTOR SAFETY SYSTEMS - Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action. Manual protective action is considered part of the reactor safety system.

1.27 REACTOR SECURED - The reactor is secured when:

1. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderator and reflection, or

2. The following conditions exist:

   a. All control devices (3 control rods) are fully inserted;
   b. The console key switch is in the off position and the key is removed from the lock;
   c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods; and
   d. No experiments in or near the reactor are being moved or serviced that have, on movement, the smaller of: a reactivity worth exceeding the maximum value allowed for a single experiment, or a reactivity of one dollar.

1.28 REACTOR SHUTDOWN - The reactor is shut down if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments included and the following conditions exist:

   a. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods;

b. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

1.29 REFERENCE CORE CONDITION - The reference core condition is the reactivity condition of the core when it is at 20 °C and the reactivity worth of xenon is zero (i.e., cold, clean, and critical).

1.30 REPORTABLE OCCURRENCE - A reportable occurrence is any of the following:

1. Operation with actual safety-system settings for required systems less conservative than the Limiting Safety-System Settings specified in technical specifications 2.2.

2. Operation in violation of the Limiting Conditions for Operation established in the technical specifications.

3. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests. (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required performs their intended reactor safety function.)

4. An unanticipated or uncontrolled change in reactivity greater than one dollar.

5. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable.

6. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

1.31 ROD-CONTROL - A control rod is a device fabricated from neutron absorbing material which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod may be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.32 SAFETY CHANNEL - A safety channel is a measuring channel in the reactor safety system.

1.33 SAFETY LIMIT - Safety limits are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.

1.34 SCRAM TIME - Scram time is the elapsed time between the initiation of a scram signal by either automated or operator initiated action and the time required for the control rods to reach a fully inserted position into the core.

1.35 SECURED EXPERIMENT - A secured experiment is any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces with are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

1.36 SECURED SHUTDOWN - Secured shutdown is achieved when the reactor meets the requirements of the definition of "reactor secured" and the facility administrative requirements for leaving the facility with no licensed reactor operators present.

1.37 SENIOR REACTOR OPERATOR - A senior reactor operator (SRO) is an individual who is licensed by the NRC to direct the activities of reactor operators.

1.38 SHALL, SHOULD, MAY – The word 'shall' is used to denote a requirement; the word 'should' is used to denote a recommendation; and the word 'may' is used to denote permission, neither a requirement nor a recommendation.

1.39 SHUTDOWN MARGIN - Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operation condition and with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.40 SHUTDOWN REACTIVITY - Shutdown reactivity is the value of the reactivity of the reactor with all control rods in their least reactive position (e.g., inserted). The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

1.41 STANDARD CORE - A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

1.42 STEADY STATE MODE - Steady state mode operation shall mean operation of the reactor with the mode selector switch in the STEADY STATE position.

1.43 TRUE VALUE - The true value is the actual value of a parameter.

1.44 UNSCHEDULED SHUTDOWN - An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions which could adversely affect safe operation, not to include shutdowns which occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

2.1 SAFETY LIMIT

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding will result.

Specification

The temperature in a standard TRIGA fuel element shall not exceed 1000 °C under any conditions of operation, with the fuel fully immersed in water.

Basis

The important parameter for TRIGA reactor is the UZrH fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium. The data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of ZrH_x will remain below the ultimate stress if the temperature in the fuel does not exceed 1000 °C and the fuel cladding is water-cooled.

It has been shown by experience that operation of TRIGA reactors at a power level of 1000 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500 kW. Analysis and measurements on other TRIGA reactors have shown that a power level of 1000 kW corresponds to a peak fuel temperature of approximately 400 °C.

2.2 LIMITING SAFETY SYSTEM SETTINGS

Applicability

This specification applies to the reactor scram setting that prevents the reactor fuel temperature from reaching the safety limit.

Objective

The objective is to provide a reactor scram to prevent the safety limit (fuel element temperature of 1000 °C) from being reached.

Specification

The limiting safety system setting shall be 175 °C as measured by the instrumented fuel element. The instrumented element may be located at any position in the core.

Basis

A Limiting Safety Setting of 175 °C provides a safety margin of 650 °C. A part of the safety margin is used to account for the difference between the temperature at the hot spot in the fuel and the measured temperature resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the thermocouple element is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations have shown that if the thermocouple element were located on the periphery of the core, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, when the temperature in the thermocouple element reaches the setting of 175 °C, the true temperature at the hottest location would be no greater than 350 °C, providing a margin to the safety limit of at least 650 °C. This margin is ample to account for the remaining uncertainty in the accuracy of the fuel temperature, measurement channel, and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR CORE PARAMETERS

Applicability

These specifications shall apply to the reactor at all times it is operating.

Objective

The objectives are to ensure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specifications

1. The excess reactivity relative to the cold critical conditions, with or without experiments in place shall not be greater than $3.50.

2. The shutdown margin shall not be less than $0.50 with:

   a. The reactor in the reference core condition; and

   b. Total worth of all in-core experiments in their most reactive state; and

   c. Most reactive control rod fully withdrawn.

3. Core configurations:

   a. The reactor shall only be operated with a standard core.

   b. No fuel shall be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive fuel element.

   c. No control rods shall be removed from the core unless a minimum of four fuel bundles are removed from the core.

   d. The reactor shall be operated only with three operable control rods.

4. No operation with damaged fuel (defined as a clad defect that results in fission product release into the reactor coolant) except to locate such fuel.

5. The reactivity coefficients for the reactor are:

Fuel:	-1.2 ¢/ °C
Moderator:	+3.0 ¢/ °C
Power:	-0.53 ¢/kW

6. The burnup of U-235 in the UZrH fuel matrix shall not exceed 50 % of the initial concentration.

Bases

1. While specification 3.1.1, in conjunction with specification 3.1.2, tends to over constrain the excess reactivity, it helps ensure that the operable core is similar to the core analyzed in the FSAR.

2. The value of the shutdown margin as required by specification 3.1.2 assures that the reactor can be shutdown from any operating condition even if the highest worth control rod should remain in the fully withdrawn position.

3. Specification 3.1.3 ensures that the operable core is similar to the core analyzed in the FSAR. It also ensures that accidental criticality will not occur during fuel or control rod manipulations.

4. Specification 3.1.4 limits the fission product release that might accompany operation with a damaged fuel element. Fuel will be considered potentially "Damaged" if said fuel is found to be leaking under the air and/or water sampling or under such case that the fuel has been exposed to temperature above 175 °C as measured on the instrumented fuel element. The criteria of the water and air sampling to determine a leaking fuel rod is considered positive if either sample is found to contain I-129 through I-135, Xe-135, Kr-85, 87 and Kr-88, Cs-135 and Cs-137, or Sr-89 through Sr-92.

5. The reactivity coefficients in Specification 3.1.5 ensure that the net reactivity feedback is negative.

6. General Atomic tests of TRIGA fuel indicate that keeping fuel element burnup below 50 % of the original ^{235}U loading will avoid damage to the fuel from fission product buildup.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

Applicability

These specifications apply to reactor control and safety systems and safety-related instrumentation that must be operable when the reactor is in operation.

Objective

The objective of these specifications is to specify the lowest acceptable level of performance or the minimum number of operable components for the reactor control and safety systems.

Specifications

1. The drop time of each of the three standard control rods from the fully withdrawn position to the fully inserted position shall not exceed one second.

2. Maximum positive reactivity insertion rate by control rod motion shall not exceed $0.30 per second.

3. The reactor safety channels shall be operable in accordance with Table 3.1, including the minimum number of channels and the indicated maximum or minimum set points for the scram channels.

4. The safety interlocks shall be operable in accordance with Table 3.2, including the minimum number of interlocks.

5. The Beam Port and Through Tube interlocks may be bypassed during a reactor operation with the permission of the Reactor Director.

6. A minimum of one reactor power channel, calibrated for reactor thermal power, must be attached to a recording device sufficient for auditing of reactor operation history.

Bases

1. Specification 3.2.1 assures that the reactor will be shutdown promptly when a scram signal is initiated. Experiments and analysis have indicated that for the range of transients anticipated for the MUTR TRIGA reactor, the specified control rod drop time is adequate to assure the safety of the reactor.

2. Specification 3.2.2 establishes a limit on the rate of change of power to ensure that the normally available reactivity and insertion rate cannot generate operating conditions that exceed the Safety Limit. (See FSAR)

3. Specification 3.2.3 provides protection against the reactor operating outside of the safety limits. Table 3.3 describes the basis for each of the reactor safety channels.

4. Specification 3.2.4 provides protection against the reactor operating outside of the safety limits. Table 3.4 describes the basis for each of the reactor safety interlocks.

5. Specification 3.2.5 ensures that reactor interlocks will always serve their intended purpose. This purpose is to assure that the operator is aware of the status of both the beam ports and the through tube.

6. Specification 3.2.6 provides for a means to monitor reactor operations and verify that the reactor is not operated outside of its license condition.

Table 3.1: Reactor Safety Channels: Scram Channels

Scram Channel	Minimum Required Operable	Scram Setpoint
Reactor Power Level	2	Not to exceed 120 %
Fuel Element Temperature	1	Not to exceed 175 °C
Reactor Power Channel Detector Power Supply	2	Loss of power supply voltage to chamber
Manual Scram	1	N/A
Console Electrical Supply	1	Loss of electrical power to the control console
Rate of power change – Period	1	Not less than 5 seconds

Table 3.2: Reactor Safety Channels: Interlocks

Interlock/Channel	Function
Log Power Level	Provide signal to period rate and minimum source channels. Prevent control rod withdrawal when neutron count rate is less than 1cps.
Startup Count rate	Prevent control rod withdrawal when neutron count rate is less than 1 cps.
Safety 1 Trip Test	Prevent control rod withdrawal when Safety 1 Trip Test switch is activated.
Plug Electrical Connection	Disable magnet power when Beam Port or Through Tube plug is removed unless bypass has been activated.
Rod Drive Control	Prevent simultaneous manual withdrawal of two or more control rods in the steady state mode of operation.

Table 3.3: Reactor Safety Channels: Scram Channel Bases

Scram Channel	Bases
Reactor Power Level Fuel Element Temperature	Provides protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded.
Reactor Power Channel Detector Power Supply	Provides protection to assure that the reactor cannot be operated unless the neutron detectors that input to each of the linear power channels are operable.
Manual Scram	Allows the operator to shut down the reactor if an unsafe or abnormal condition occurs.
Console Electric Supply	Assures that the reactor cannot be operated without a secure electric supply.
Rate of power change – Period	Assures that the reactor is operated in a manner that allows the operator time to shut down the reactor before the licensed power restriction is exceeded.

Table 3.4: Reactor Safety Channels: Interlock Bases

Interlock/Channel	Bases
Log Power Level	This channel is required to provide a neutron detector input signal to the start up count rate channel.
Startup Count rate	Assures sufficient amount of startup neutrons are available to achieve a controlled approach to criticality.
Safety 1 Trip Test	Assures that the 1 cps interlock cannot be bypassed by creating an artificial 1 cps signal with the Safety 1 trip test switch
Plug Electrical Connection	Assures that the reactor cannot be operated with Beamport or Through Tube plugs removed without further precautions.
Rod Drive Control	Limits the maximum positive reactivity insertion rate available for steady state operation.

3.3 COOLANT SYSTEMS

Applicability

This specification applies to the quality and quantity of the primary coolant in contact with the fuel cladding at the time of reactor startup.

Objectives

1. To minimize the possibility for corrosion of the cladding on the fuel elements.
2. To minimize neutron activation of dissolved materials.
3. To ensure sufficient biological shielding during reactor operations.
4. To maintain water clarity.

Specifications

1. A minimum of 15 ft. of coolant shall be above the core.
2. Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm and the pH shall be between 5.0 and 7.5. Conductivity shall be measured before each reactor operation. pH shall be measured monthly, interval not to exceed six weeks.
3. Gross gamma measurement shall be less than two times historical data measurements. Gross gamma activity shall be measured monthly, interval not to exceed six weeks.
4. The pool water temperature shall not exceed 40 C.

Bases

1. Specification 3.3.1 ensures that both sufficient cooling capability and sufficient biological shielding are available for safe reactor operation.
2. A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limit provides acceptable control. In addition, by limiting the concentration of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operation.
3. Specification 3.3.3 ensures that a fuel failure with release of radioactive materials into the pool will be determined.
4. Specification 3.3.4 ensures a DNBR value greater than 2.

3.4 CONFINEMENT

Applicability

This specification applies to that part of the facility that contains the reactor, its controls and shielding.

Objective

The objective of these specifications is to ensure that sufficient confinement volume is available for the dilution of radioactive releases.

Specifications

1. Confinement shall be considered established when the doors leading from the reactor bay area leading into the balcony area on the top floor, and the reception area as well as the building exterior are secured.

2. Confinement shall be established whenever the reactor is in an unsecured mode with the exception of the time that persons are physically entering or leaving the confinement area.

Bases

These specifications will dilute and delay the release of radioactive materials and ensure that the release conditions are similar to those assumed in the SAR.

3.5 VENTILATION SYSTEMS

Applicability

These specifications apply to the ventilation systems for the reactor building.

Objective

The objective of these specifications is to ensure that air exchanges between the reactor confinement building and the environment do not impact negatively on the general public.

Specifications

1. Air within the reactor building shall not be exchanged with other occupied spaces in the building.

2. All locations where ventilation systems exchange air with the environment shall have failsafe closure mechanisms.

3. Forced air ventilation to the outside shall automatically secure without operator intervention in such case that the radiation levels exceed a preset level as defined in facility procedures. The setpoints are: 50 mR/hr (bridge monitor), 10 mR/hr (exhaust monitor).

Bases

1. This specification ensures that radioactive releases inside the reactor building will not be transported to the remainder of the building.

2. This specification ensures that the reactor building can always be isolated from the environment.

3. This specification ensures that radioactive release will be minimized by stopping forced flow to the outside environment.

3.6 RADIATION MONITORING SYSTEM AND EFFLUENTS

3.6.1 Radiation Monitoring System

Applicability

This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specifications

1. The reactor shall not be operated unless a minimum of one of the two radiation area monitor channels listed in Table 3.5 are operable.

2. For a period of time not to exceed 48 hours for maintenance or calibration to the radiation monitor channels, the intent of specification 3.6.1 will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be observable by the reactor operator.

3. The alarm set points shall be stated in a facility operating procedure. The alarm setpoints for the bridge monitor are: 37 mR/hr (alert), 50 mR/hr (scram). The setpoints for the exhaust monitor are: 8 mR/hr (alert), 10 mR/hr (scram).

4. The campus radiation safety organization shall maintain an environmental monitor at the MUTR site boundary.

Table 3.5: Minimum Radiation Monitoring Channels

Radiation Area Monitors	Function	Minimum Number Operable
Exhaust Radiation Monitor	Monitor radiation levels in Reactor Bay area at an Exhaust Fan location	A minimum of 1 of the 2 monitors shall be operable
Bridge Radiation Monitor	Monitor radiation levels in Reactor Bay area at the Reactor Bridge location	

Bases

Specification 3.6.1 ensures that a significant fuel failure with release of radioactive materials will be determined and that any large releases will be mitigated by the specified protective actions.

The radiation area monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

The alarm and scram set points shall be designed to ensure that dose rates delivered to areas accessible to members of the general public do not exceed the levels defined in 10 CFR Part 20.

The additional function of the radiation area monitor that monitors the reactor bay area is to warn personnel entering the building of high radiation levels if the pool water level should decrease to the level of inadequate biological shielding.

The intent of 3.6.3 and 3.6.4 are to ensure that facility does not lead to a dose to the general public greater than that allowed by 10 CFR Part 20.

3.6.2 Effluents

Applicability

This specification applies to limits on effluent release.

Objective

The objective is to ensure that the releast of radioactive materials from the reactor facility to unrestricted areas do not exceed federal regulations.

Specification

All effluents from the MUTR shall conform to the standards set forth in 10 CFR Part 20.

Basis

The intent of 3.6.2 is to ensure that, in the event that radioactive effluents are released, the dose to the general public will be less than that allowed by 10 CFR Part 20.

3.7 LIMITATIONS ON EXPERIMENTS

Applicability

The specification applies to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive material in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist.

1. The reactivity worth of any single experiment shall be less than $1.00.

2. The total absolute reactivity worth of in-core experiments shall not exceed $3.00 , including, the potential reactivity which might result from experimental malfunction and experiment flooding or voiding.

3. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated.

4. Explosive materials in quantities greater than 25 mg TNT or its equivalent shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be-less than the design pressure of the container. The failure pressure of the container is one half of the design pressure.Total explosive material inventory in the reactor facility may not exceed 100 mg TNT or its equivalent.

5. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor or (3) possible accident conditions in the experiment shall be limited in activity such that if 100 % of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor room or ouside environment will not result in exceeding the applicable dose limits set forth in 10 CFR Part 20.

   In calculations pursuant to 3.7.5 above, the following assumptions shall be used:

   a. If the effluent from an experimental facility exhausts through a holdup tank, which closes automatically on high radiation level, at least 10 % of the gaseous activity or aerosols produced will escape.

   b. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99 % efficiency for 0.3 μm particles, at least 10 % of these particles can escape.

c. If an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100 per cent of the radioactive gases or aerosols escape.

d. If an experiment fails that contains materials with a boiling point above 130° F (54° C), the vapors of at least 10 percent of the materials escape through an undisturbed column of water above the core.

6. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 5 mCi.

Bases

1. This specification is intended to provide assurance that the worth of a single unsecured experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were to be inserted suddenly.

2. The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since experiments of such worth must be fastened in place, its inadvertent removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained.

   The maximum worth of all experiments is also limited to a reactivity value such that the cold reactor will not achieve a power level high enough to exceed the core temperature safety limit if the experiments were removed inadvertently.

3. This specification is intended to prevent damage to reactor components resulting from experiment failure. If an experiment fails, inspection of reactor structures and components shall be performed in order to verify that the failure did not cause damage. If damage is found, appropriate corrective actions shall be taken.

4. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials, especially the accidental detonation of the explosive. If an experiment fails, inspection of reactor structures and components shall be performed in order to verify that the failure did not cause damage. If damage is found, appropriate corrective actions shall be taken.

5. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Table 11 of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

6. The 5 mCi limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area. (See SAR)

4.0 SURVEILLANCE REQUIREMENTS

INTRODUCTION

Surveillances shall be performed on a timely basis as defined in the individual procedures governing the performance of the surveillance. In the event that the reactor is not in an operable condition, such as during periods of refueling, or replacement or repair of safety equipment, surveillances may be postponed until such time that the reactor is operable. In such case that any surveillance must be postponed, a written directive signed by the Facility Director, shall be placed in the records indicating the reason why and the expected completion date of the required surveillance. This directive shall be written before the date that the surveillance is due. Under no circumstance shall the reactor perform routine operations until such time that all surveillances are current and up to date. Any system or component that is modified, replaced, or had maintenance performed will undergo testing to ensure that the system/component continues to meet performance requirements.

4.1 REACTOR CORE PARAMETERS

Applicability

These specifications apply to the surveillance requirements for the reactor core.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.1 are satisfied.

Specifications

1. The excess reactivity shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of control rods.

2. The shutdown margin shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of control rods

3. Core configuration shall be verified prior to the first startup of the day.

4. Gross gamma measurements shall be taken monthly, at intervals not to exceed six weeks.

5. Twenty percent of the fuel elements shall be inspected visually annually such that the entire core is inspected over a five year period.

6. Burnup shall be verified in the Annual Report.

Bases

Experience has shown that the identified frequencies ensure performance and operability for each of these systems or components. For excess reactivity and shutdown margin, long-term changes are slow to develop. For fuel inspection, visually inspecting 20% of the bundles annually will identify any developing fuel integrity issues throughout the core.

4.2 REACTOR CONTROL AND SAFETY SYSTEMS

Applicability

These specifications apply to the surveillance requirements for reactor control and safety systems.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.2 are satisfied.

Specifications

1. The reactivity worth of each standard control rod shall be determined annually, intervals not to exceed 15 months, and after each time the core fuel configuration is changed or a control rod is changed.

2. The control rod withdrawal and insertion speeds shall be determined annually, intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect rod travel times.

3. Control rod drop times shall be measured annually; intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect their drop time.

4. All scram channels and power measuring channels shall have a channel test, including trip actions with safety rod release and specified interlocks performed after each secured shutdown, before the first operation of the day, or prior to any operation scheduled to last more than 24 hours, or quarterly, with intervals not to exceed 4 months. Scram channels shall be calibrated annually, intervals not to exceed 15 months.

5. Operability tests shall be performed on all affected safety and control systems after any maintenance is performed.

6. A channel calibration shall be made of the linear power level monitoring channels annually, intervals not to exceed 15 months.

7. A visual inspection of the control rod poison sections shall be made biennially, intervals not to exceed 28 months.

8. A visual inspection of the control rod drive and scram mechanisms shall be made annually, intervals not to exceed 15 months.

Bases

1. The reactivity worth of the control rods, specification 4.2.1, is measured to assure that the required shutdown margin is available and to provide a means to measure the reactivity worth of experiments. Long term effects of TRIGA reactor operation are such that measurements of the reactivity worths on an annual basis is adequate to insure that no significant changes in shutdown margin have occurred.

2. The control rod withdrawal and insertion rates, specification 4.2.2, are measured to insure that the limits on maximum reactivity insertion rates are not exceeded.

3. Measurement of the control rod drop time, specification 4.2.3, ensures that the rods can perform their safety function properly.

4. The surveillance requirement specified in specification 4.2.4 for the reactor safety scram channels ensures that the overall functional capability is maintained.

5. The surveillance test performed after maintenance or repairs to the reactor safety system as required by specification 4.2.5 ensures that the affected channel will perform as intended.

6. The linear power level channel calibration specified in specification 4.2.6 will assure that the reactor will be operated at the licensed power levels.

7. Specification 4.2.7 assures that a visual inspection of control rod poison sections is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.

8. Specification 4.2.8 assures that a visual inspection of control drive mechanisms is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.

4.3 COOLANT SYSTEMS

Applicability

These specifications apply to the surveillance requirements of the reactor coolant systems.

Objective

The objective of these specifications is to ensure the operability of the reactor coolant system as described in Section 3.3.

Specifications

1. The primary coolant level shall be verified before each reactor startup or daily during operations exceeding 24 hours.

2. Pool water conductivity shall be determined prior to the first startup of the day, and pool water pH shall be determined monthly at intervals not to exceed six weeks.

3. Pool water gross gamma activity shall be determined monthly, at intervals not to exceed six weeks. If gross gamma activity is high (greater than twice historical data), gamma spectroscopy shall be performed.

4. Pool water temperature shall be measured prior to the reactor startup and shall be monitored during reactor operation.

Bases

1. Gross gamma activity measurements are conducted to detect fission product releases from damaged fuel element cladding.

2. Specification 4.3.2 ensures that poor pool water quality could not exist for long without being detected. Years of experience at the MUTR have shown that pool water analysis on a monthly basis is adequate to detect degraded conditions of the pool water in a timely manner.

3. Specification 4.3.3 ensures that sufficient water exists above the core to provide both sufficient cooling capacity and an adequate biological shield.

4. Specification 4.3.4 ensures that the maximum allowable pool water temperature is not exceeded.

4.4 CONFINEMENT

Applicability

This specification applies to that part of the facility which contains the reactor, its controls and shielding.

Objective

The objective of these specifications is to ensure that radioactive releases from the confinement can be limited.

Specifications

Prior to a reactor startup the isolation of the confinement building shall be visually verified.

Bases

This specification ensures that the minimal leakage rate assumed in the SAR is actually present during reactor operations in order to limit the release of radioactive material to the environs.

4.5 VENTILATION SYSTEM

Applicability

This specification applies to the reactor ventilation system.

Objective

The objective is to assure that provisions are made to restrict the amount of radioactivity released to the environment.

Specification

The ability to secure the ventilation system shall be verified before each reactor startup.

Bases

The facility is designed such that in the event that excessive airborne radioactivity is detected the ventilation system shall be shutdown to minimize transport of airborne materials. Analysis indicates that in the event of a major fuel element failure personnel would have sufficient time to evacuate the facility before the maximum permissible dose (10 CFR Part 20) is exceeded.

4.6 RADIATION MONITORING SYSTEM AND EFFLUENTS

4.6.1 Radiation Monitoring System

Applicability

This specification applies to the surveillance requirements for the Radiation Area Monitoring System (RAMS).

Objective

The objective of these specifications is to ensure the operability of each radiation area monitoring channel as required by Section 3.6 and to ensure that releases to the environment are kept below allowable limits.

Specifications

1. A channel calibration shall be made for each channel listed in Table 3.5 annually but at intervals not to exceed 15 months or whenever maintenance or repairs are made that could affect their calibration.

2. A channel test shall be made for each channel listed in Table 3.5 prior to starting up the reactor to ensure reactor scram, fan shutdown, and louver closing.

Bases

Specifications 4.6.1.1 and 4.6.1.2 ensure that the various radiation area monitors are checked and calibrated on a routine basis, in order to assure compliance with 10 CFR Part 20.

4.6.2 Effluents

Applicability

This specification applies to the surveillance requirements for air and water effluents.

Objective

The objective of these specifications is to that releases to the environment are kept below allowable limits.

Specifications

1. Reactor building air samples shall be counted for gross gamma activity monthly, intervals not to exceed 6 weeks.

2. A sample of any water discharged from the reactor building sump shall be counted for gross gamma activity before its release to the environs.

Bases

Specifications 4.6.2.1 and 4.6.2.2 ensure that the facility effluents comply with 10 CFR Part 20.

4.7 EXPERIMENTS

Applicability

This specification applies to the surveillance requirements for experiments installed in the reactor and its irradiation facilities.

Objective

The objective of this specification is to prevent the conduct of experiments which may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specifications

1. The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with said experiment

2. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.7 by the Reactor Safety Committee (new experiment) or Facility Director (modified routine experiment), in full accord with Section 6.2.3 of these Technical Specifications and the procedures which are established for this purpose.

Basis

Experience has shown that experiments reviewed and approved by the Reactor Safety Committee or Facility Director can be conducted without endangering the safety of the reactor, personnel, or exceeding Technical Specification Limits.

5.0 DESIGN FEATURES

5.1 SITE CHARACTERISTICS

Applicability

This specification applies to the reactor facility and its site boundary.

Objective

The objective is to assure that appropriate physical security is maintained for the reactor facility and the radioactive materials contained within it.

Specifications

1. The reactor shall be housed in a closed room designed to restrict leakage. The closed room does not include the West balcony area.
2. The reactor site boundary shall consist of the outer walls of the reactor building and the area enclosed by the loading dock fence.
3. The restricted area shall consist of all areas interior to the reactor building including the west balcony and lower entryway.
4. The controlled area shall consist of all areas interior to the reator building including the west balcony and lower entryway.

Bases

These specifications assure that appropriate control is maintained over access to the facility by members of the general public.

5.2 REACTOR COOLANT SYSTEM

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

1. The reactor core shall be cooled by natural connective water flow.

2. The pool water inlet pipe is equipped with a siphon break at the surface of the pool.

3. The pool water return (outlet) pipe shall not extend more than 50.8 cm (20 in) below the overflow outlet pipe when fuel in the core.

Bases

Specification 5.2.1 is based on thermal and hydraulic calculations and operation of other TRIGA reactors that show that a core can operate in a safe manner at power levels up to 1500 kW with natural convection flow of the coolant.

Specification 5.2.2 and 5.2.3 ensures that the pool water level can normally decrease only by 50.8 cm (20 in) if the coolant piping where to rupture and siphon water from the reactor tank. Thus, the core will be covered by at least 4.57 m (15 ft) of water.

5.3 REACTOR CORE AND FUEL

Applicability

This specification applies to the configuration of the core and in-core experiments.

Objective

The objective is to ensure that the core configuration is as specified in the license.

Specifications

1. The core shall consist of 93 TRIGA fuel elements assembled into 24 fuel bundles – 21 bundles shall contain four fuel elements and 3 bundles shall contain three fuel elements and a control rod guide tube.

2. The fuel bundles shall be arranged in a rectangular 4 x 6 configuration, with one bundle displaced for the in-core pneumatic experimental system.

3. The reactor shall not be operated at power levels exceeding 250 kW.

4. The reflector shall be a combination of two graphite reflector elements and water

Basis

1. Only TRIGA fuel elements shall be used in the fuel bundles.

2. The experimental system allows insertion of small samples directly into the reactor core.

3. The maximum power level presents a conservative limitation with respect to the safety limits for the maximum temperature in the fuel.

5.3.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics, and that the fuel used in the reactor has characteristics consistent with the fuel assumed in the SAR and the license.

Specifications

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

1. Uranium content: a maximum of 9.0 w/o uranium enriched to less than 20 % ^{235}U
2. Zirconium hydride atom ratio: nominal 1.5 - 1.8 hydrogen-to-zirconium, ZrH_x
3. Cladding: 304 stainless steel, nominal thickness of 0.508 mm (.020 in)
4. The overall length of a fuel element shall be 30 inches, and the fueled length shall be 15 inches.

Basis

The design basis of the standard TRIGA fuel element demonstrates that 250 kW steady state operation presents a conservative limitation with respect to safety limits for the maximum temperature generated in the fuel.

5.3.2 Control Rods

Applicability

This specification applies to the control rods used in the reactor core.

Objective

The objective is to assure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

1. The three control rods shall have scram capability, shall be used for reactivity control, and shall contain borated graphite, B_vC, in powder form.
2. The control rod cladding shall be aluminum with nominal thickness 0.71 mm (0.028") and length 17".

Basis

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C, powder. These materials must be contained in a suitable clad material such as aluminum to ensure mechanical stability during movement and to isolate the poison from the tank water environments. Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor.

5.4 FISSIONABLE MATERIAL STORAGE

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications

1. All fuel elements shall be stored either in a geometrical array where the k-effective is less than 0.8 for all conditions of moderation and reflection or stored in an approved fuel shipping container.

   Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

2. When fuel is in storage in any area other than the grid plate, that area must be equipped with monitoring devices that both measure and record the radiation levels and temperature of the region surrounding the fuel.

Basis

The limits imposed by Specifications 5.4.1 and 5.4.2 are conservative and assure safe storage.

6.0 ADMINISTRATION

6.1 ORGANIZATION

The Maryland University Training Reactor (MUTR) is owned and operated by the University of Maryland, College Park. Its position in the university's structure is shown in Figure 6.1

The university shall provide whatever resources are required to maintain the facility in a condition that poses no hazard to the general public or to the environment.

6.1.1 Structure

Figure 6.2 shows the MUTR organizational structure.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility and radiological safety shall rest with the Facility Director. The members of the organization chart shown in Figure 6.2 shall be responsible for safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the operating license.

6.1.3 Facility Staff Requirements

1. The minimum staffing when the reactor is operating shall be:

    a. A licensed reactor operator (RO) or a licensed senior reactor operator (SRO) shall be present in the control room.

    b. A minimum of two persons shall be present in the facility or in the Chemical and Nuclear Engineering Building when the reactor is operating: the operator in the control room and a second person who can be reached from the control room who is able to carry out prescribed written instructions which may involve activating elements of the Emergency Plan, including evacuation and initial notification procedures.

    c. A licensed SRO shall be present or readily available on call. "Readily Available on Call" means an individual who (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where he/she may be rapidly contacted and the method of contact, and (3) is capable of arriving at the reactor facility within a reasonable amount of time under normal conditions. At no time shall the designated SRO be more than thirty minutes or ten miles from the facility.

2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:

    a. Management personnel

    b. Radiation safety personnel

c. Licensed operator

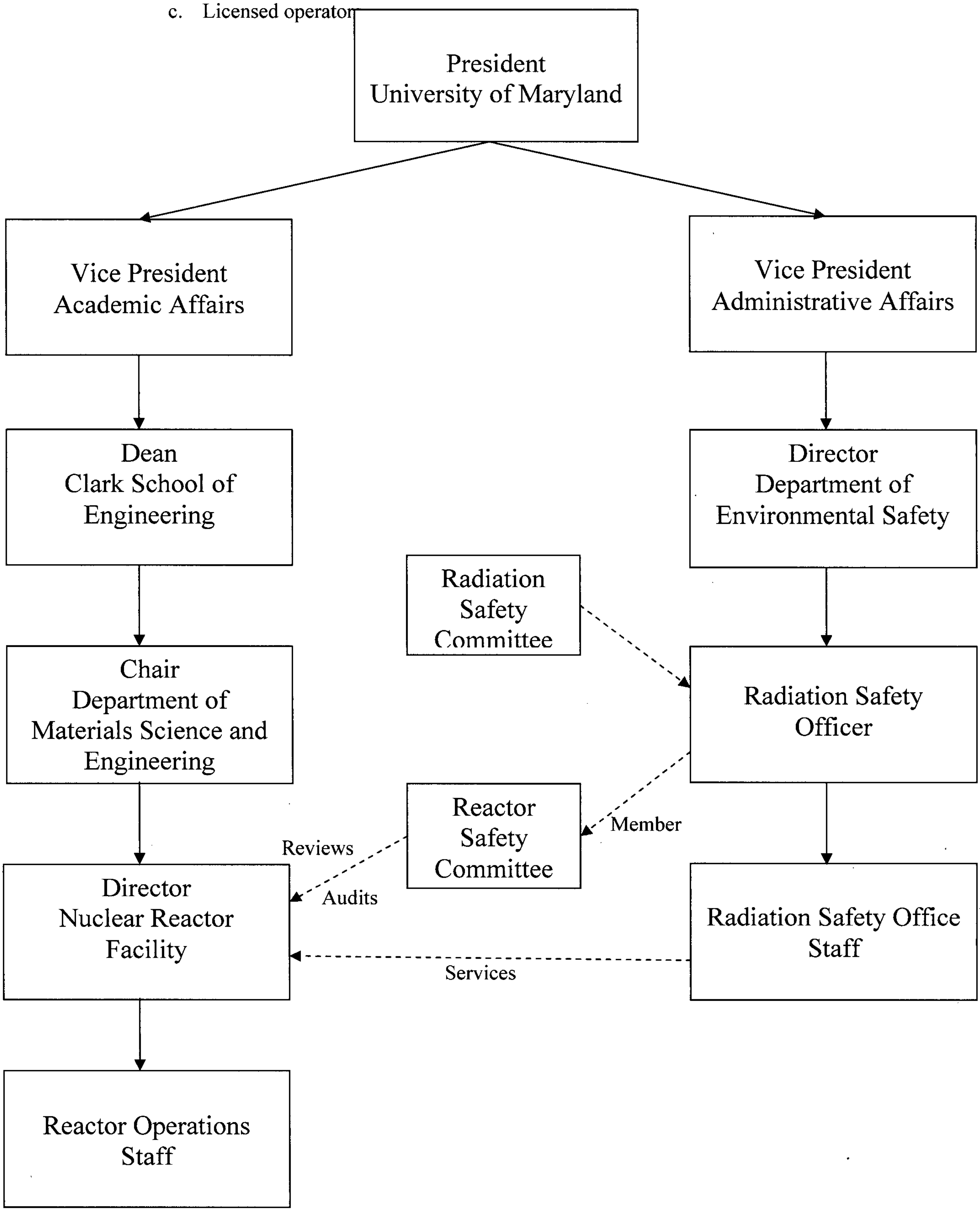

Figure 6.1: MUTR Position in University of Maryland Structure

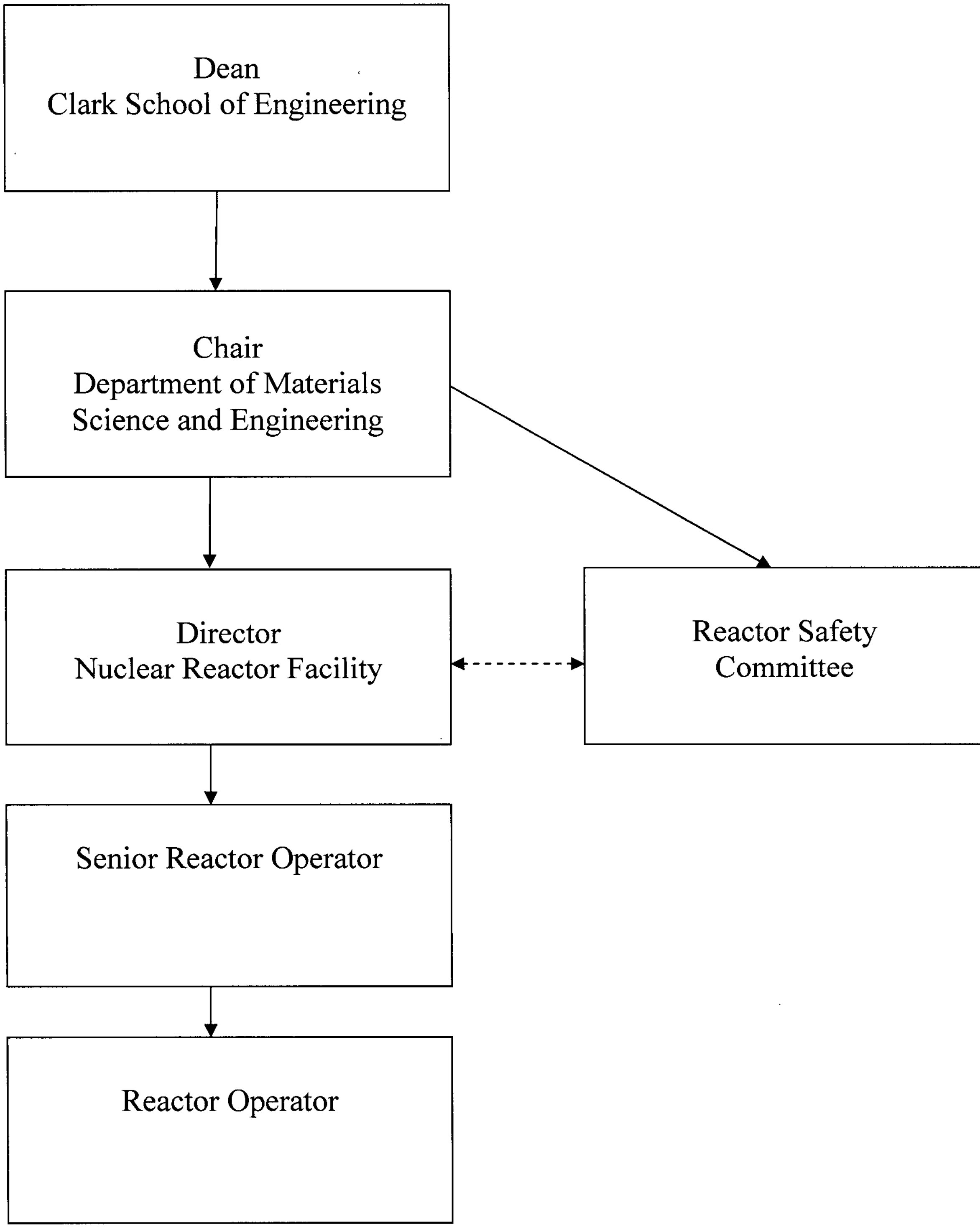

Figure 6.2: MUTR Organizational Structure

3. The following operations shall be supervised by a senior reactor operator:

    a. Initial startup and approach to power following new fuel loading or fuel rearrangement

    b. When experiments are being manipulated in the core that have an estimated worth greater than $0.80

    c. Removal of control rods or fuel manipulations in the core

    d. Resumption of operation following an unplanned or unscheduled shutdown or any unplanned or unexpected significant reduction in power.

6.1.4 Selection and Training of Personnel

The selection and training of operations personnel should be in accordance with the following:

1. Responsibility - The Facility Director or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators. This selection shall be in conjunction with the guidelines set forth in ANSI/ANS 15.1 and 15.4.

6.2 REVIEW AND AUDIT

6.2.1 Reactor Safety Committee

A Reactor Safety Committee (RSC) shall exist for the purpose of reviewing matters relating to the health and safety of the public and facility staff and the safe operation of the facility. It is appointed by and reports to the Chairperson of the Department of Materials Science and Engineering. The RSC shall consist of a minimum of five persons with expertise in the physical sciences and preferably some nuclear experience. Permanent members of the committee are the Facility Director and the Campus Radiation Safety Officer or that office's designated alternate, neither may serve as the committee's chairperson. Qualified alternates may serve on the committee. Alternates may be appointed by the Chairperson of the RSC to serve on a temporary basis. At least one committee member must be from outside the Department of Materials Science and Engineering.

6.2.1.1 Reactor Safety Committee Charter And Rules

1. The RSC shall meet at least twice per year, and more often as required.

2. A quorum of the RSC shall be not less than half of the committee members, one of whom shall be the Campus Radiation Safety Officer (or designated alternate). No more than two alternates shall be used to make a quorum. MUTR staff members shall not constitute the majority of a voting quorum.

3. Minutes of all meetings will be retained in a file and distributed to all RSC members.

6.2.1.2 Reactor Safety Committee Review Function

The RSC shall review the following:

1. Determinations that proposed changes in equipment, systems, test, experiments, or procedures are allowed without prior authorization by the responsible authority, e.g. 10 CFR 50.59;

2. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance;

3. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Proposed changes in technical specifications, or license;

5. Violations of technical specifications or license. Violations of internal procedures or instructions having safety significance;

6. Operating abnormalities having safety significance;

7. Reportable occurrences listed in Section 6.7.2;

8. Audit reports.

A written report of the findings and recommendations of the RSC shall be submitted to Level 1 management, the Facility Director, and the RSC members in a timely manner after the review has been completed.

6.2.1.3 Reactor Safety Committee Audit Function

1. An annual audit and review of the reactor operations will be performed by an outside individual or group familiar with research reactor operations. They shall submit a report to the Facility Director and the Reactor Safety Committee.

2. The following shall be reviewed:

    a. Reactor operators and operational records for compliance with internal rules, procedures, and regulations, and with license provisions;

    b. Existing operating procedures for adequacy and accuracy;

    c. Plant equipment performance and its surveillance requirements;

    d. Records of releases of radioactive effluents to the environment;

    e. Operator training and requalification;

f. Results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety; and
g. Reactor facility emergency plan and implementing procedures.

Deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management and the Facility Director. A written report of the findings of the audit shall be submitted to Level 1 management, the Facility Director, and the RSC members within 3 months after the audit has been completed.

6.2.2 Audit Of ALARA Program

The Facility Director or his designated alternate shall conduct an audit of the reactor facility ALARA Program at least once per calendar year (not to exceed fifteen months). The results of the audit shall be presented to the RSC at the next scheduled meeting. This audit may occur as part of a review of the overall campus ALARA program.

6.3 RADIATION SAFETY

A radiation safety program following the requirements established in 10 CFR Part 20 will be undertaken by the Radiation Safety Office. The facility director will ensure that ALARA principles are followed during all facility activities.

6.4 OPERATING PROCEDURES

Written procedures, reviewed and approved by the Reactor Safety Committee, shall be in effect and followed for the following items prior to performance of the activity. The procedures shall be adequate to assure the safety of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

1. Start-up, operation, and shutdown of the reactor

2. Installation or removal of fuel elements, control rods, experiments, and experimental facilities

3. Maintenance procedures that could have an effect on reactor safety

4. Periodic surveillance checks, calibrations, and inspections required by the Technical Specifications or those that may have an effect on reactor safety

5. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity

6. For any activity pertaining to shipping, possession, and transfer of radioactive material, these procedures shall be written in conjunction with the Radiation Safety Office and the Radiation Safety Officer who shall inform the Reactor Director of any changes in regulations or laws that may require modification of these procedures. All shipping and receiving of radioactive material shall be performed in conjunction with, and with the approval of the Radiation Safety Office.

7. Implementation, maintenance, and modification to the Emergency Plan
8. Implementation, maintenance, and modification to the Security Plan
9. Implementation, maintenance, and modification to the Radiation Protection Plan. The Radiation Protection Plan shall include an ALARA plan as defined in ANSI/ANS-15.11

10. Use, receipt, and transfer of byproduct material

Substantive changes to the above procedures shall be made with the approval of the Facility Director and the Reactor Safety Committee and shall be made in accordance with 10 CFR 50.59. This approval shall be granted before the changes may be considered in effect. The only exception to this clause is in such a case where the delay in implementation would cause a credible risk to the public or the facility. If such a case exists as determined by the Facility Director, temporary approval may be granted by the Director but must be approved by the Reactor Safety Committee within thirty days. Temporary or minor changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Committee at the next scheduled meeting. The Reactor Director shall have the power to approve minor changes such as phone number changes, typographical error correction or any other change that does not change the effectiveness or the intent of the procedure. It shall be considered sufficient approval and documentation when the Director forwards by electronic means to both the Radiation Safety Officer and the Chair of the Reactor Safety Committee. A copy of the transmission shall be filed with the appropriate procedure.

6.5 EXPERIMENT REVIEW AND APPROVAL

1. Routine experiments may be performed at the discretion of the duty senior reactor operator without the necessity of further review or approval.

2. Modified routine experiments shall be reviewed and approved in writing by the Facility Director, or designated alternate.

3. Special experiments shall be reviewed by the RSC and approved by the RSC and the Facility Director or desired alternate prior to initiation.

4. The review of an experiment listed in subsections 6.5.2 and 6.5.3 above, shall consider its effect on reactor operation and the possibility and consequences of its failure, including, where significant, chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and any reactivity effects.

6.6 REQUIRED ACTIONS

6.6.1 Actions To Be Taken In Case Of Safety Limit Violation

In the event a safety limit is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

2. The event shall be reported to the Reactor Director who will report to the NRC as required in section 6.7.2.

3. An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.7.2 of these specifications, and

4. A report, and any followup report, shall be prepared. The report shall describe the following:

   a. Applicable circumstances leading to the violation, including when known, the cause, and contributing factors;

   b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and
   c. Corrective action to be taken to prevent recurrence.

   The report shall be reviewed by the Reactor Safety Committee and submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Actions to Be Taken In The Event Of a Reportable Occurrence

In the event of a reportable occurrence, as defined in section 1.27 of these Technical Specifications, the following actions will be taken:

1. Immediate action shall be taken to correct the situation and to mitigate the consequences of the occurrence.

2. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the Facility Director.

3. The event shall be reported to the Facility Director who will report to the NRC as required in section 6.7.2.

4. The Reactor Safety Committee shall investigate the causes of the occurrence at its next meeting. The Reactor Safety Committee shall report its findings to the NRC and Dean, School of Engineering. The report shall include an analysis of the causes of the occurrence, the effectiveness of corrective actions taken, and recommendations of measures to prevent or reduce the probability or consequences of recurrence.

6.7 REPORTS

6.7.1 Annual Operating Report

A report summarizing facility operations shall be prepared annually for the reporting period ending June 30. This report shall be submitted by September 30 of each year to the NRC Document Control Desk. The report shall include the following:

1. A brief narrative summary of results of reactor operations and surveillance tests and inspections required in section 4.0 of these Technical Specifications

2. A tabulation showing the energy generated in MW hr^{-1} for the year

3. A list of unscheduled shutdowns including the reasons therefore and corrective action taken, if any

4. A tabulation of the major maintenance operations performed during the period, including the effects, if any, on safe operation of the reactor, and the reason for any corrective maintenance required

5. A brief description of

    a. Each change to the facility to the extent that it changes a description of the facility in the Final Safety Analysis Report

    b. Review of changes, tests, and experiments made pursuant to 10 CFR Part 50.59.

6. A summary of the nature and amount of radioactive effluents released or discharged to the environment

7. A description of any environmental surveys performed outside of the facility

8. A summary of exposure received by facility personnel and visitors where such exposures are greater than 25 percent of limits allowed by 10 CFR Part 20

9. Changes in facility organization

6.7.2 Special Reports

Notification shall be made within 24 hours by telephone to the NRC Operations Center, followed by a written report faxed within 14 days in the event of the following:

1. A reportable occurrence, as defined in Section 1.27 of this document

2. Release of radioactivity from the site above allowed limits

3. Exceeding the Safety Limit

The written report shall be sent to the NRC document control desk. The written report and, to the extent possible, the preliminary telephone or facsimilie notification shall:

1. Describe, analyze, and evaluate safety implications

2. Outline the measures taken to ensure that the cause of the condition is determine

3. Indicate the corrective action taken to prevent repetition of the occurrence including chances to procedures

4. Evaluate the safety implications of the incident in light of the cumulative experience obtained from the report of previous failure and malfunction of similar systems and components

6.7.3 Unusual Event Report

A written report shall be forwarded within 30 days to the NRC Document Control Desk, with a copy to the Regional Administrator, Region I, NRC, in the event of:

1. Discovery of any substantial errors in the transient or accident analysis or in the methods used for such analysis as described in the Safety Analysis Report or in the bases for the Technical Specifications

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications or Safety Analysis Report

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed failure, could result in a loss of the capability of the system to perform its safety function

4. A permanent change in the position of Department Chair or Facility Director

6.8 RECORDS

1. The following records shall be retained for a period of at least five years:

   a. Normal reactor facility operation and maintenance

   b. Reportable occurrences

   c. Surveillance activities required by Technical Specifications

   d. Facility radiation and contamination surveys

   e. Experiments performed with the reactor

   f. Reactor fuel inventories, receipts, and shipments

   g. Approved changes in procedures required by these Technical Specifications

   h. Minutes of the Reactor Safety Committee meetings

   i. Results of External Audits

2. Retraining and requalification records of current licensed operators shall be maintained at all times that an operator is employed or until the operator's license is renewed.

3. The following records shall be retained for the lifetime of the facility:

   a. Liquid radioactive effluents released to the environs

b. Gaseous radioactive effluents released to the environs

c. Radiation exposure for all facility personnel
d. Radiation exposures monitored at site boundary

e. As-built facility drawing

f. Violation of the Safety Limit

g. Violation of any Limited Safety System Setting (LSSS)

h. Violation of any Limiting Condition of Operation (LCO)

4. Requirement 6.8.1 (a) above does not include supporting documents such as checklists, logsheets and recorder charts, which shall be maintained for a period of at least one year.

5. Applicable annual reports, if they contain any of the required information may be used as records in subsection 6.8.3 above.

Responses to RAI Questions of 2-18-2011 Meeting

RAI-4
You proposed to provide a response to this RAI by adding a technical specification for annual visual inspection of 20% of core fuel elements. A different sample will be chosen each year. This will result in a complete core inspection in five years.

The requirement for visual inspection of 20% of the fuel elements annually (complete inspection over a five year period) has been added as TS 4.1.5

RAI-6
You proposed to provide a response to this RAI by adding an LCO for the maximum pool temperature based on the day to day operations of MUTR. In response to an earlier question on the reactor coolant (RAI-19) you had stated that a limit of 40 C on the pool temperature was set to avoid damage to the ion exchange resin that is used to maintain water conductivity within acceptable limits.

The GA calculation on pool temperature terminates at approximately 93 C due to calculation instabilities. There is a general concern on the impact of cycling stresses from low to high pool temperatures on the aluminum pool liners fatigue cracking and resulting leaks. This has already occurred at one facility, the University of Wisconsin. See their response to RAI in ADAMS Accession Number ML101690137. As described in the regulations 10CFR 50.36(a)(2), LCOs are the lowest functional capability or performance levels of equipment required for safe operation of a facility. An example of these LCOs are identified in both AFRRI and Dow TRIGA facilities TS (ML041800068 and ML092150443 respectively)

A pool water temperature LCO has been set at 40 C (TS 3.3.4). This value is based on DNBR calculations from GA, and ensures a DNBR > 2.

RAI-8, RAI-10 and RAI-12
You proposed to respond to this RAI by providing radiological analyses discussing dose calculations; the first with the facility ventilation on and the second with the ventilation off. The two scenarios of interest are (i) MHA and (ii) routine operations. For each of these scenarios you will calculate and provide values for (a) occupational dose, (b) dose to maximally exposed member of the public and (c) dose to exposed member of the public located in the areas adjacent to the MUTR (classroom, outer wall, etc.). In our discussions, you stated that as part of your analysis, you will describe how you will meet 10CFR20 as part of your routine effluent discharge and whether the operation of the facility ventilation system is required during normal operations and accident situations.

An example is identified in a response from the University of Wisconsin, ADAMS Accession Number ML091110549.

Refer to report of calculations done by University of Maryland Radiation Safety Office.

RAI-23
IFE calibration method should test whether a signal is generated when the setpoint is reached. This could be accomplished by an simulated electronic signal to the circuitry. Calibration at low, equilibrium temperature does not seem to test the scram circuit.

A calibration check is part of the Startup Checklist (OP 101), which is performed prior to the first startup of the day. Two electronic signals are sent to the circuitry – one to check zero (0 C ± 5 C) and high temperature (500 C ± 10 C). The scram circuit is checked to ensure that a scram signal is generated. After checking 0C and 500C, the console is reset, and the input electronic signal is increased from a low value until a scram signal is generated, which must be at a temperature ≤ 175 C.

Non-TS RAI-2
You proposed to provide the calculation of the departure from nucleate boiling ratio (DNBR) for the hottest location in the MUTR core. You informed the NRC that GA should be able to provide you with this information.

See the response to RAI 6.

Revisions to RAI Responses Submitted 31 January 2011
Original response in blue, revisions in red.

4. TS 3.1.4: Please define fuel damage. ANSI/ANS-15.1-2007, Section 3.1(6) indicates that limits shall be established for fuel inspections. Please discuss how MUTR inspects fuel elements and under which conditions is fuel considered damaged.

Fuel damage is defined as a clad defect that results in fission product release into the reactor coolant.

General Atomic published a report (GA-A16613) in 1981 which details an investigation of fuel damage found in the Texas A&M reactor. The reactor has a maximum power level of 1 MW and can pulse. Over approximately a three year period (June 1973 to September 1976), the core operated 287 MWd in steady state and pulsed 725 times. The maximum pulse insertion was $2.70, with a corresponding peak core temperature rise of 883° C. During a loading operation in September 1976, four 'somewhat deformed' (terminology from the report) fuel elements were seen. These fuel elements were in the closest proximity to the transient rod throughout their operating history. Inspection of fuel elements in the next lower flux region showed no damage.

Pulsing operations were subsequently suspended, and no additional fuel damage was noted. The report concludes that the damage was due to pulsing, and the steady-state history of the fuel is not a factor.

The report noted that fuel inspection at the University of Wisconsin and Washington State University reactors, each with pulsing capability, showed no fuel damage.

Routine inspection of MUTR fuel has never been required. As a conversion TRIGA, an inspection of a fuel element would require mechanical disassembly of the four fuel element assembly.

MUTR is low power (250 kW), cannot pulse, and typical burnup is about 1 MWd per year. Based on the assessment of the Texas A&M fuel damage, and the conclusions reached in the GA report, a routine fuel inspection is not required. As noted, if fuel damage (cladding defect) occurs, fission products would be released into the reactor coolant, and these would be detected in the pool water gamma analysis (TS 4.3.1).

Additional Response: A requirement for visual fuel inspection has been added as TS 4.1.5

5. TS 3.2: The applicability statement of the TS needs to be labeled. Minimum channels needed for operation appear to be missing from the TS. Please address. In Table 3.2 clarify the log power level and explain the interlock.

'<u>APPLICABILITY</u>' will be added above the first sentence in TS 3.2. A new table (Table 3.5) will be added which lists minimum channels required for operation. Minimum channels required for operation are listed in Table 3.1. In table 3.2, the 'function' description of the log power level will be changed to indicate the interlock functions of the channel.

Additional Response: Table 3.1 has been modified to list minimum channels required for operation. Therefore, there is no need for an additional table.

6. TS 3.3: ANSI/ANS-15.1-2007, Section 3.3(9) indicates that limits shall be established for water chemistry requirements. MUTR TS 4.3.5 includes pH and conductivity values, which are considered LCO limits and should be moved to TS 3.3. Please discuss whether the LCO conditions in TS 4.3.5 should be placed in TS 3.3 and also include an LCO for gross gamma measurements or justify why it is not needed. Should an LCO be established for maximum pool water temperature? The numbering of the water (coolant) specifications and their bases should be made consistent.

   The pH and conductivity LCO values will be deleted from TS 4.3.5 and added to TS 3.3. An LCO for gross gamma measurement will be added. This LCO will be 'a gross gamma measurement that is more than two times greater than historical data measurements'.

   MUTR is designed and licensed as a natural circulation system (SAR 4.6). At all power levels, fuel is cooled via natural circulation and the heat is convected from the pool to the reactor building atmosphere. Therefore, there is no requirement for an LCO based on coolant water temperature.

   The numbering of the coolant specifications and bases will be revised for consistency.

   Additional Response: A pool temperature LCO has been added as TS 3.3.4.

12. TS 3.5.1: Does this TS mean that the reactor confinement is airtight? If not, please explain airflow pathways to the atmosphere during operation and emergency conditions. Please revise the TS to reflect the operation of the system.

   The reactor confinement is not airtight. During normal operation, if the ventilation fans are running, there is a pathway to the outside. If high radiation levels are detected, the ventilation fans automatically shut down and the louvers close, which minimizes release to the outside environment. Additionally, the confinement design limits any release to other occupied spaces in the building. There are no ventilation fans that exhaust from the reactor confinement into the building, and the doors into the reactor confinement area are required to be closed during operation. TS 3.5.1 will be revised to reflect these conditions.

   Additional Response: TS 3.5.1 as written explains operation of the system. Thus, no revisions are needed.

13. TS 3.5.3: Explain the automatic operation of securing the forced air ventilation. Securing is interpreted as turning off the fans and other components. What are the preset radiation levels for securing ventilation? How are facility personnel protected by securing the forced air ventilation?

Fans are turned off and louvers closed to secure the system. Preset levels are 37 mR/hr (alert), 50 mR/hr (scram) for the bridge monitor and 8 mR/hr (alert), 10 mR/hr (scram) for the exhaust monitor. Securing the forced air ventilation is not designed to protect facility personnel, but rather to minimize uncontrolled radioisotope release to the outside environment.

Additional Response: Calculations (provided as part of response to RAI's of 18 February) indicate that personnel doses are within 10 CFR 20 limits with forced air ventilation secured.

18. TS 3.7.4: This specification states in part that explosive materials in quantities less than 25 mg TNT may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the containment. Section 10.3 of the UMTR SAR states that calculations must show that the pressure produced if detonation occurs is less than the failure pressure of the container. Since the container design pressure should have a safety factor of two (Regulatory Guide 2.2), the failure pressure should be half the design pressure. Therefore, TS 3.7.4 should be modified accordingly. Moreover, there are no example calculations in the SAR comparing the detonation pressure to the failure pressure. Please provide an example calculation for a container that demonstrates compliance with the factor of 2 margin or justify not including a calculation in the SAR.

This specification is now TS 3.7.5 (in the revised version). The statement 'The failure pressure of the container is one half of the design pressure.' will be added after '….less than the design pressure of the container'

A discussion and analysis (attached) will be added in the MUTR SAR. This is taken from the Oregon State University reactor SAR.

Additional Response: The number of the specification (TS 3.7.4) has not changed.

23. TS 4.2.4. The TS refers to the calibration of scram channels. There appears to be no calibration required for the instrumented fuel element which measures fuel temperatures. Please add a surveillance requirement or explain why it is not needed.

A surveillance requirement for the IFE will be added to TS 4.2.4. Calibration will be done using coolant temperature as the reference value (once the system has reached equilibrium with the reactor shut down).

Additional Response: Please see response of 18 February.

Response to the NRC request for further information regarding the dose to workers, members of the public and the nearest residence during normal operations and during an MHA incident at the UMTR, University of Maryland Training Reactor.

Sources of radiation during normal and MHA conditions:

During normal operations in the reactor facilities, Ar-41 is produced primarily from irradiation of dissolved argon gas in the reactor pool, a fraction of which eventually evolves into the air of the reactor bay. This evolution results from the reduced solubility of argon gas in water as the water temperature increases. Additionally, Ar-41 can be generated from activated argon in air filled cavities such as the thermal column, beam-ports, and through-tube. These auxiliary systems are utilized in approximately 1% of experiments and therefore contribute a negligible amount of Ar-41 to the reactor bay in comparison to the pool generation and diffusion from the pool to the air volume in the reactor.

The UMTR is capable of exhausting the air in the reactor bay via two exhaust fans mounted on the side of the reactor building. The fans exhaust at a rate of 2.83 m^3/s. During normal operations the fans are not in operation. Two conditions exist for the normal operation and MHA event: Ventilation ON (fans running), and Ventilation OFF (fans turned off). In the case of Ventilation OFF, a conservative assumption is made that there will be leakage at a rate of 5% per hour out of the reactor bay through several cavities such as the entrance/exit doors on the ground floor and upper balcony level as well as the roll up door on the ground floor of the reactor bay.

The Maximum Hypothetical Accident, MHA that could occur as stated in the SAR is a fuel element manipulation incident that results in the failure of one fuel element in air with subsequent release of isotopes into the reactor bay volume. Values of the release of isotopes are listed in tables 13.1-13.3 of the SAR. There are 6 conditions that are considered in regards to doses to the operator, nearby member of the public and nearest residence for both an MHA and for normal operations.

During an MHA and during Normal Operations:

1. Occupational dose to workers ventilation ON
2. Occupational dose to workers ventilation OFF
3. Maximum dose to the public at the nearest residence ventilation ON
4. Maximum dose to the public at the nearest residence ventilation OFF
5. Maximum dose to a member of the public adjacent to MUTR ventilation ON
6. Maximum dose to a member of the public adjacent to MUTR ventilation OFF

Action taken, assumptions, formulas, and reference data are as follows:

- The production rate as stated in the SAR for Ar-41 in the reactor pool during normal operations is 0.1 Ci / 30 MWhrs operation. This rate is utilized to determine the steady state equilibrium rate of Ar-41 generation in the pool for 250 kw, 365 days, 24 hours per day for a whole year
- Total Effective Dose Equivalents (TEDE) is the sum of external dose as well, (DDE) and the internal Committed Effective Dose Equivalent (CEDE);
- Building Leakage was assessed at 12 different locations in the Reactor building utilizing Drager smoke emitting air current tubes to observe the flow and direction of smoke in the reactor bay;
- A leakage rate of .05/hr was assumed out of the reactor bay with ventilation OFF;
- Uniform mixing and instantaneous release of the nuclides during an MHA was assumed;
- Dose Conversion Factors (DCFs) reflect those contained in IAEA Safety Series No. 115, Federal Guidance Report 11 and ICRP 68 for the isotopes in question;
- The Occupational Dose to workers is taken over 2000 hours;
- The Gaussian Plume model and Pasquill Categories of Atmospheric Stability were utilized to determine maximum concentration downwind of the reactor for doses to the nearest residence;

DOSE CALCULATIONS

- <u>The Occupational internal dose to workers:</u>

$$D = C \bullet (Br) \bullet t \bullet (DCF)$$

Where:

D = Dose to the worker inside the reactor [rem]
C = concentration $C_{v,}$ C_L ventilation and leakage [Ci/m^3]
Br = breathing rate [m^3/s]
DCF = Dose Conversion Factor [rem/Ci]
t = time to evacuate the building [s]

- <u>The Occupational external dose to workers:</u>

$$D\gamma \text{ [rads/s]} = (0.057)\, E_{\gamma,\,avg}\,(\text{MeV})\,\chi\ [\text{Ci/m}^3]$$

Where:

$D\gamma$ = Gamma dose rate from the cloud in [rads/s]

$E_{\gamma,\,avg}$ = Gamma ray energy in MeV
χ = the concentration in the cloud in [Ci/m^3]

$$D_\beta = 0.23 E_{avg} \chi \text{ [rads/s]}$$

Where:

D_β = Beta dose rate from the cloud in [rads/s]
E_{avg} = Average Beta ray in MeV
χ = the concentration in the cloud in [Ci/m^3]

- The maximum dose to the nearest residence:

Using the Gaussian Plume model for the ground level centerline concentration:

$$\chi = \frac{Q}{2\pi\sigma_y\sigma_z\mu} \text{ EXP}\left(-\frac{y^2}{2\sigma_y^2}\right) \left[\text{EXP}\left(\frac{-(z-h)^2}{2\sigma_z^2}\right) + \text{EXP}\left(\frac{-(z+h)^2}{2\sigma_z^2}\right)\right]$$

For an elevated release, ventilation ON (fans running), and a center line concentration:

$$\chi = \frac{Q}{2\pi\sigma_y\sigma_z\mu} \; e^{-\frac{H^2}{2\sigma_z^2}}$$

For a ground release, ventilation OFF (leakage), and a center line concentration:

$$\chi = \frac{Q}{2\pi\sigma_y\sigma_z\mu}$$

The maximum ground level concentration occurs for $\sigma_z = H/\sqrt{2}$

Where H is defined as the effective height:

$$H = h + d \cdot (v/\mu)^{1.4}(1+\Delta T)/T$$

Where h = effective release height in meters
d = diameter of the outlet in meters

v = exit velocity of the gas m/s
μ = average wind speed m/s
ΔT = difference in the ambient and effluent gas temperature
T = temperature of the effluent gas

$$D = f_{rel} A \bullet (\chi/Q) \bullet (Br) \bullet (DCF)$$

Where:

D = Dose to the public outside the reactor [rem]
f_{rel} = fraction of activity released, f_v, f_L ventilation and leakage
A = total activity that could be released [Ci]
X/Q = atmospheric diffusion factor [s/m^3]
Br = breathing rate [m^3/s]
DCF = Dose Conversion Factor [rem/Ci]

- The maximum dose to person in areas adjacent to MUTR:

Condition: a member of the public located on the lower level outside the reception room on the south side of the reactor, figure 3.4 of the SAR: in the case of ventilation ON, the only source of exposure to the individual would be from shine inside the reactor bay. In the case of ventilation OFF the individual could receive an exposure from leakage out of the reactor and shine from source terms in the reactor.

Summary of doses:

MHA Summary

Occupational dose to workers ventilation ON = 219 mrem
Occupational dose to workers ventilation OFF = 278 mrem

Max dose to public (nearest residence) ventilation ON = 28.6 μrem
Max dose to public (nearest residence) ventilation OFF = 39.4 μrem

Max dose rate to public at door to reactor ventilation ON = 1.22 mrem/hr
Max dose rate to public at door to reactor ventilation OFF = 25.86 mrem/hr

Normal Operations Summary

The occupational dose to the worker with ventilation OFF assumes that 100% of the Ar-41 readily diffuses out of the water and into the reactor volume. Air and Water sampling during operation at full power has shown that the majority of the Ar-41 remains in the pool and is not diffused to the atmosphere, therefore a factor of at least ¼ can be applied to the activity in the reactor bay due to Ar-41 production in the pool. For 2000 hours of operation the dose to workers is:

Occupational dose to workers ventilation ON = 19.6 mrem
Occupational dose to workers ventilation OFF = 289 mrem

For a member of the public located on the lower level outside the double doors on the south side of the reactor in the case of ventilation ON, the only source of exposure to the individual would be from shine inside the reactor bay. In the case of ventilation OFF the individual could receive an exposure from leakage out of the reactor and from any shine from Ar-41 production inside the reactor.

Annual dose to the public at the door to the reactor, ventilation ON = 3.71 mrem

Annual dose to the public at the door to the reactor, ventilation OFF will be determined by air sampling and quantification of Ar-41 over the pool and in the reactor bay volume, with subsequent dose calculation. The reactor is currently not operating in order to install new fission chamber elements. Upon start up Ar-41 will be collected by grab sample and measured through HpGe and NaI gamma spectroscopy.

Doses to the public located at the nearest residence from elevated releases (ventilation ON) and from a ground release (ventilation OFF) is calculated using the Gaussian Plume model.

Max dose to public (nearest residence) ventilation ON = 6.33 mrem
Max dose to public (nearest residence) ventilation OFF = 0.200 mrem